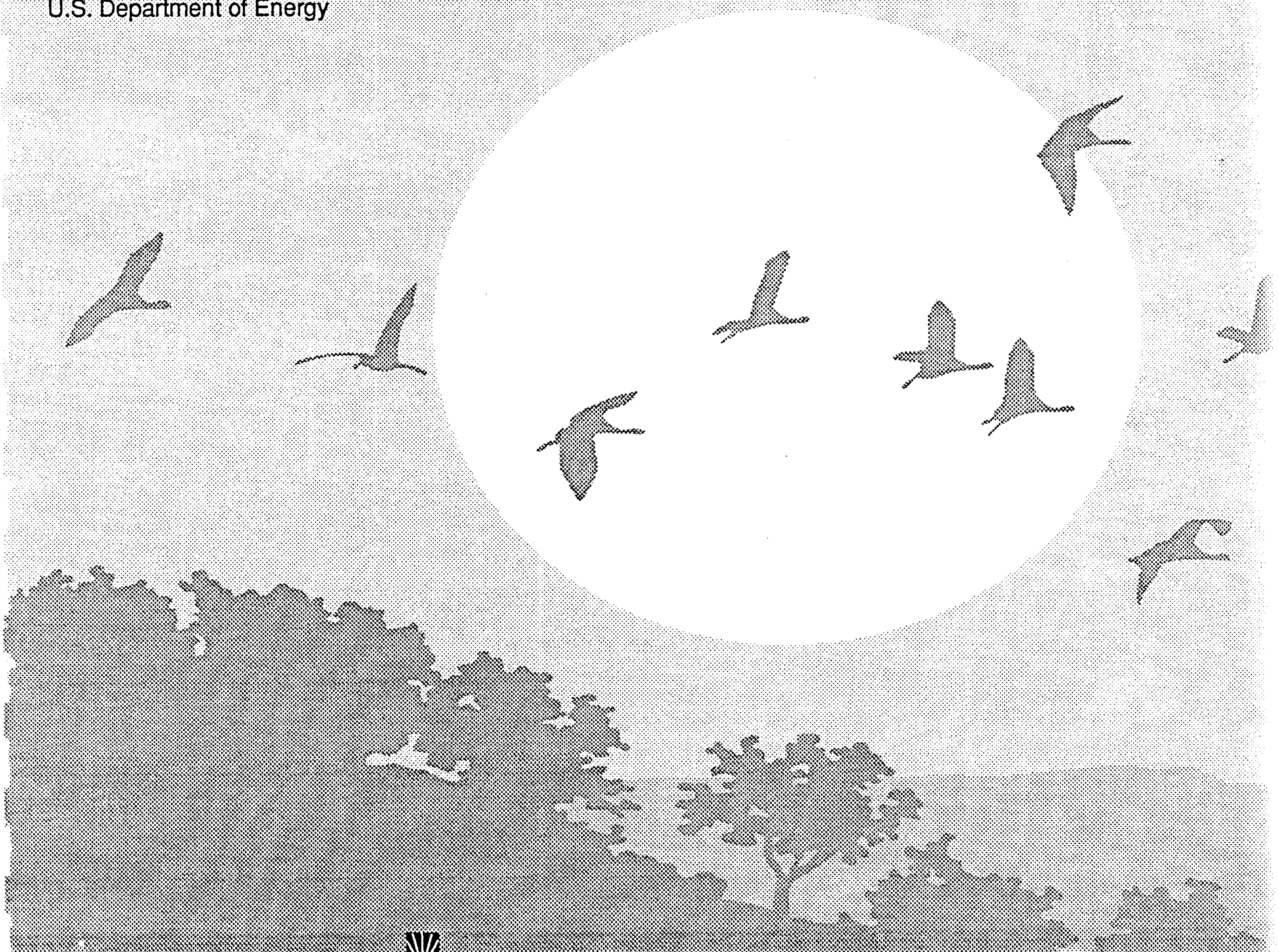
RADIOLOGICAL SURVEY OF THE BUILDING 059 REACTOR VAULT SANTA SUSANA FIELD LABORATORY ROCKWELL INTERNATIONAL VENTURA COUNTY, CALIFORNIA

T. J. VITKUS AND J. R. MORTON

Prepared for the
Office of Environmental Restoration
U.S. Department of Energy



ORISE

OAK RIDGE INSTITUTE FOR SCIENCE AND EDUCATION

Environmental Survey and Site Assessment Program
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SANTA SUSANA FIELD LABORATORY
ROCKWELL INTERNATIONAL
VENTURA COUNTY, CALIFORNIA

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TABLE OF CONTENTS

LIST OF FIGURES

LIST OF TABLES

ABBREVIATIONS AND ACRONYMS

AEC	Atomic Energy Commission
cm^2	square centimeter
cm^3	cubic centimeter
DOE	Department of Energy
dpm/100 cm^2	disintegrations per minute per 100 square centimeters
ERDA	Energy Research and Development Administration
ESSAP	Environmental Survey and Site Assessment Program
ETEC	Energy Technology Engineering Center
GM	Geiger-Mueller
ha	hectare
m	meter
mR/h	milliroentgens per hour
ORISE	Oak Ridge Institute for Science and Education
pCi	picocuries
pCi/g	picocuries per gram
pC/l	picocuries per liter
PCR	pipe chase room
PIC	pressurized ionization chamber
SNAP	Systems for Nuclear Auxiliary Power
SSFL	Santa Susana Field Laboratory

RADIOLOGICAL SURVEY OF THE BUILDING 059 REACTOR VAULT SANTA SUSANA FIELD LABORATORY ROCKWELL INTERNATIONAL VENTURA COUNTY, CALIFORNIA

INTRODUCTION AND SITE HISTORY

Rockwell International's Rocketdyne Division operates the Santa Susana Field Laboratory (SSFL). The Energy Technology Engineering Center (ETEC) is that portion of the SSFL, operated for the Department of Energy (DOE), which performs testing of equipment, materials, and components for nuclear and energy related programs. Contract work for the Atomic Energy Commission (AEC) and the Energy Research and Development Administration (ERDA), predecessor agencies to the DOE, began in the early 1950's. Specific programs conducted for AEC/ERDA/DOE involved the engineering, development, testing, and manufacturing operations of nuclear reactor systems and components. Other SSFL activities have also been conducted for the National Aeronautics and Space Administration, the Department of Defense, and other government related or affiliated organizations and agencies. Some activities have been licensed by the Nuclear Regulatory Commission and by the State of California Radiological Health Branch of the Department of Health Services.

Numerous buildings and land areas became radiologically contaminated as a result of the various operations which included ten reactors, seven criticality test facilities, fuel fabrication, reactor and fuel disassembly, laboratory work, and on-site storage of nuclear material. Potential radioactive contaminants identified at the site are uranium (in natural and enriched isotopic abundances), plutonium, americium-241, fission products (primarily cesium-137 and strontium-90), activation products (tritium [H-3], cobalt-60 [Co-60], europium-152 [Eu-152], europium-154 [Eu-154], nickel-63, promethium-147, tantalum-182). Chemical contaminants, mainly chlorinated organic solvents, have also been identified in groundwater, primarily as a result of rocket engine testing.

Decontamination and decommissioning of contaminated facilities began in the late 1960's and continues as other DOE sponsored projects are phased out and transitioned to DOE EM-40.

Rockwell/Rocketdyne is currently completing the decommissioning and final status radiological surveys of Building 059.

Building 059 was constructed during 1962 and 1963 to house a research and development program known as the Systems for Nuclear Auxiliary Power (SNAP). The program was established to develop and test power systems for use as subsidiary nuclear power units in specialized applications. Atomic International developed the SNAP 8 reactor, under contract to the AEC, for use in this program. Reactor operation began in 1963 followed by a shutdown in 1964 in order to permit building modifications. These modifications included rebuilding the north reactor test cell and construction of an additional underground structure. The new area accommodated a vacuum system, consisting of a vacuum equipment room and a pipe chase room, used for the simulation of a space environment within the north test cell during reactor tests. The last test was conducted in 1969 at which time the reactor was shut down. The SNAP project resulted in the radiological contamination of those portions of Building 059 associated with reactor operations. Contamination was principally the result of neutron activation; the primary contaminants were Co-60, Eu-152, Eu-154, iron-55 (Fe-55), and H-3. Decommissioning of the facility was initiated in the 1970's followed by annual inspections and radiological survey work. The only remaining radioactivity was in the concrete of the reactor test cell and the pipe chase room. Inspections performed in 1983 revealed groundwater leaking into the facility. Rockwell/Rocketdyne stabilized the problem and a water management plan was put into place. Structural deterioration was revealed in 1987 when water was discovered in previously dry areas. Due to the potential risk of contamination by way of pathways between the outside and the vault interior, the Building 059 Vault Remediation Program was initiated.

To date, the decontamination and decommissioning work on the SNAP reactor vault has resulted in the removal and disposal of all equipment and material associated with the SNAP Test Program. In addition, the test cell concrete walls and floor have been scabbled to the first layer of reinforcement steel (rebar). Although decommissioning efforts have reduced the levels of residual contamination, further remediation of cell walls to eliminate residual activity would compromise the building's structural integrity. As a result, Rockwell/Rocketdyne has requested approval of guidelines for the reactor test cell that are based on a pathway analysis using

RESRAD.[1,2] The pathway analysis Rockwell/Rocketdyne performed assumes a final building configuration where the test cell and the adjoining pipe chase room have been backfilled with concrete, thereby rendering the area inaccessible.

DOE's Office of Environmental Restoration, Northwestern Area Programs is responsible for oversight of a number of remedial actions that have been or will be conducted at the SSFL. It is the policy of DOE to perform independent (third party) verification of remedial action activities conducted within Office of Environmental Restoration programs. The purpose of these independent verifications is to confirm that remedial actions have been effective in meeting established and supplemental guidelines and that the documentation accurately and adequately describes the radiological conditions at the site. The Environmental Survey and Site Assessment Program (ESSAP) of the Oak Ridge Institute for Science and Education (ORISE) was designated as the organization responsible for this task at SSFL, and was requested to verify the current radiological status of the north reactor test cell and pipe chase room, prior to backfill.

SITE DESCRIPTION

The SSFL is located in the Simi Hills of southeastern Ventura County, California, approximately 47 kilometers (29 miles) northwest of downtown Los Angeles (Figure 1). The site is comprised of approximately 1,090 hectares (ha [2,700 acres]) and is divided into four administrative areas (Areas I through IV) and a Buffer Zone. DOE operations are conducted in Rockwell International-owned and DOE-owned facilities located within the 117 ha Area IV. The ETEC portion of Area IV consists of government-owned buildings that occupy 36 ha.

Building 059 is located at the intersection of 20th and B Streets in the north central part of Area IV (Figure 2). The facility is a concrete and Butler type structure with approximately 1000 square meters of floor space. The test vault has inside dimensions of 8.5 meters (m) wide, 12 m long, by 9.8 m deep. The top of the test vault ceiling is at ground level and serves as the floor of the high bay area. The two test cells are located below the west end of the vault floor and are identified as the north and south reactor cells (Figure 3).

The north reactor test cell is located at the northwest corner of the test vault and has inside dimensions of 3.6 m wide, 3.7 m long, and 4.4 m deep (Figure 4). The walls and floor are fabricated of steel-reinforced concrete with 1% boron. The adjoining pipe chase room (PCR) is 4 m high. The floor and the walls of the PCR are constructed of steel-reinforced concrete (Figure 5). The fourth wall of the PCR (east wall) is the original west wall of the test vault and reactor containment cell. The south reactor test cell was not used for testing and was not subject to activation.

The tops of the north test cell and PCR are approximately on level with the test vault floor, which is 11 m below grade. The floor of the test cell (and PCR) is approximately 15 m below grade. The test cell south wall, and most of the west wall, was removed during decommissioning, and, as discussed in the introduction, the other test cell surfaces have been heavily scabbled and the equipment removed from the PCR. In the proposed final site configuration, concrete grout will be introduced into the void space of the test cell and PCR, backfilling the area up to the existing basement (vault) floor level. Radiological measurements, for release of the building to unrestricted use, will then be performed with the building in this final configuration.

OBJECTIVE

The objective of the verification process was to ensure that the data used by Rockwell/Rocketdyne for the pathway analysis accurately represented the current radiological condition of the Building 059 test cell and PCR.

DOCUMENT REVIEW

ESSAP reviewed Rockwell/Rocketdyne's supporting documentation concerning the Building 059 remediation activities and supplemental guideline request. Comments previously provided by ESSAP were addressed by Rockwell/Rocketdyne.[3,4]

PROCEDURES

ESSAP personnel visited the SSFL on December 8, 1994 and performed visual inspections and independent measurements and sampling of the reactor test cell and adjoining pipe chase room. Surveys were conducted in accordance with a plan dated December 6, 1994, submitted to and approved by the DOE.[5]

REFERENCE SYSTEM

Measurements and sampling locations were referenced to prominent building features.

SURFACE SCANS

Surface scans for beta-gamma and gamma activity were performed over portions of the test cell walls and floor in order to determine general ranges of residual activity. Scans were performed using a microrem meter and a compensated GM detector coupled to a ratemeter-scaler with audible indicator.

SURFACE ACTIVITY MEASUREMENTS

Direct measurements for total beta-gamma activity were performed at five locations on each complete wall and the floor of both the north test cell and PCR (Figures 6 and 7). Measurement locations represented the areas of maximum residual contamination as determined by surface scans. Measurements were performed using a GM detector coupled to a ratemeter-scaler.

EXPOSURE RATE MEASUREMENTS

Exposure rate measurements were performed at 1 m from the walls and floor using a pressurized ionization chamber (PIC). Location selection represented the areas of maximum direct gamma radiation as determined by surface scans. Figures 8 and 9 show measurement locations.

MISCELLANEOUS SAMPLING

Rockwell/Rocketdyne collected concrete core samples from six floor and wall locations selected by ESSAP on the basis of maximum direct radiation levels. Four cores were from the north test cell and two were from the PCR (Figures 8 and 9). Rockwell/Rocketdyne then transversely sectioned each core into 2.5 centimeter (cm [1 inch]) increments representing the wall/floor surface to 15 cm depth, 25.5 cm depth, 62.5 cm depth, and the last 2.5 cm of each core. Water samples were collected from the French drain and from one of the Building 059 groundwater monitoring wells. An additional two water samples collected by Rockwell/Rocketdyne personnel from two other monitoring wells were provided to ESSAP for analysis.

SAMPLE ANALYSIS AND DATA INTERPRETATION

Samples and data were returned to ORISE's ESSAP laboratory in Oak Ridge, Tennessee for analysis and interpretation. Concrete samples and water samples were analyzed by solid state gamma spectrometry. The primary radionuclides of interest were Co-60, Eu-152, and Eu-154. Spectra were also reviewed for any other identifiable photopeaks. Water samples were analyzed for gross alpha and gross beta activity and for H-3. Analytical results for concrete were reported in units of picocuries per gram (pCi/g). Water samples were reported in units of picocuries per liter (pCi/L). Direct measurement data were converted to units of disintegrations per minute per 100 cm^2 (dpm/100 cm^2). Exposure rates were reported in milliroentgens per hour (mR/h).

Data and sample results were then compared with the Rockwell/Rocketdyne data used for the RESRAD pathway analysis.

FINDINGS AND RESULTS

SURFACE SCANS

Gamma scans of both the north test cell and the PCR revealed continuous elevated radiation throughout the areas. Ranges of elevated direct gamma radiation were evident on the floors and walls. Surface scans identified the areas of maximum residual contamination which were marked for further investigation.

SURFACE ACTIVITY LEVELS

Total beta-gamma surface activity levels measured in the north test cell and PCR are summarized in Table 1. Total activity levels for the north test cell ranged from 32,000 to 4,400,000 dpm/100 cm^2. The range for the total activity levels in the PCR was 9,000 to 2,600,000 dpm/100 cm^2. It is believed that a significant portion of the observed activity was the result of the gamma contribution from activation products within the concrete matrix.

EXPOSURE RATES

The exposure rates at 1 m are summarized in Table 1. The exposure rates ranged from 2.6 to 5.5 mR/h in the test cell while the PCR ranged from 0.7 to 4.0 mR/h.

RADIONUCLIDE CONCENTRATIONS IN CONCRETE

The radionuclide concentrations from the 56 sections that comprised the six core samples taken in the north test cell and PCR are summarized in Table 2. Concentration ranges for the individual 56 core sections are as follows: Co-60, less than 0.4 to 3,580 pCi/g; Co-58, less than 0.3 to 157 pCi/g; Ba-133, less than 0.4 to 323 pCi/g; Eu-152, less than 0.9 to 42,700 pCi/g; Eu-154, less than 1.3 to 3,340 pCi/g; and Eu-155, less than 0.5 to 35 pCi/g.

RADIONUCLIDE CONCENTRATIONS IN WATER

Radionuclide concentrations in water are summarized in Table 3. The ranges for the four water samples are as follows: gross alpha, 3.6 to 28.3 pCi/L; gross beta, 8.4 to 21.5 pCi/L; and tritium, less than 750 to 924 pCi/L.

DISCUSSION OF RESULTS

The exposure rate measurements performed by ESSAP did not show levels in excess of those reported by Rockwell/Rocketdyne for the test cell in its current status. Rockwell/Rocketdyne reported an exposure rate range, at 1 m, of 2 to 8 mR/h. The ESSAP range was 2.6 to 5.5 mR/h.

Rocketwell/Rocketdyne did not provide data demonstrating total surface activity levels present within the test cell and PCR. Therefore, a comparison of activity levels could not be made.

In order to evaluate the ESSAP results obtained for the individual core sections and compare these results directly with the concentration levels that Rockwell/Rocketdyne used in their RESRAD pathway analysis, it was necessary to determine the average concentration of each radionuclide identified within the concrete matrix of the test cell and PCR. This was accomplished by first determining the volume of concrete represented by 2.54 cm thick sections of each wall and the floor of the test cell and PCR. For the test cell, the estimated volume for each of these sections was 1,952,700 cm^3, determined by calculating the area of the walls and floor (in cm^2) and multiplying by 2.54 cm. The total estimated radionuclide activity, in pCi, within each volume was then calculated by averaging the activity concentration for each interval, multiplying by the volume of concrete, and multiplying by the concrete density. For example, the total estimated Co-60 activity within the first 2.54 cm of the concrete remaining in the test cell was determined as follows:

3580.0 pCi/g + 638.0 pCi/g + 101.0 pCi/g + 1.16 pCi/g (MDA values used)/4 $\times$ 1,952,700 cm^3 $\times$ 2.3 g/cm^3 = 4.9×10^9 pCi.

The radionuclide activity for depth intervals that were not represented by individual samples (i.e. 17.8 cm through 25.4 cm and 30.5 cm through 63.5 cm) were interpolated. The total radionuclide activity for each interval was then summed and resulted in the following total estimated activity levels within the test cell: 1.0×10^{10} pCi of Co-60, 4.3×10^8 pCi of Co-58, 1.1×10^9 pCi of Ba-133, 8.6×10^{10} pCi of Eu-152, 1.2×10^{10} pCi of Eu-154, and 3.5×10^8

pCi of Eu-155. These activity levels were next divided by the grams of concrete currently estimated to remain in the test cell plus the amount of concrete that will be added to backfill the void space (4.7×10^8 g). The resultant average activity concentrations for the test cell were 21 pCi/g for Co-60, 0.9 pCi/g for Co-58, 2.3 pCi/g for Ba-133, 182 pCi/g for Eu-152, 25 pCi/g for Eu-154, and 0.7 pCi/g for Eu-155. These simplified calculations were confirmed for the Co-60 and Eu-152 activity levels by determining the average activity in each core by using the Best Fit of Data followed by integration to arrive at the average value. The corresponding Rockwell/Rocketdyne activity levels for the three primary radionuclides of concern were 49 pCi/g, 270 pCi/g, and 56 pCi/g for Co-60, Eu-152, and Eu-154, respectively. ESSAP did not analyze for, nor calculate the expected levels of Fe-55 and H-3 in concrete as the contribution to the external exposure dose pathway, which is the primary dose pathway once the vault has been backfilled, is inconsequential.

Activity levels within the PCR concrete were also calculated as discussed above. The estimated total activity and respective average activity concentration levels were 3.1×10^{10} pCi and 37 pCi/g for Co-60, 1.5×10^9 pCi and 1.7 pCi/g for Co-58, 9.0×10^8 pCi and 1.1 pCi/g for Ba-133, 3.3×10^{11} pCi and 391 pCi/g for Eu-152, 1.1×10^{10} pCi and 13 pCi/g for Eu-154, and 1.1×10^9 pCi and 1.3 pCi/g for Eu-155. Rockwell/Rocketdyne did not report activity levels for the PCR, therefore, a comparison could not be made.

The groundwater sample results were evaluated for evidence of radionuclides leaching from the reactor test cell. There were no gamma emitting radionuclides detected by gamma spectrometry and gross alpha and gross beta activity levels did not indicate the presence of elevated activity. Three of the four groundwater sample H-3 levels were less than the MDAs of the procedure, which ranged from 750 to 830 pCi/L. One sample contained 920 pCi/L. The Rockwell/Rocketdyne reported H-3 level in groundwater was 500 pCi/L. After considering the random error associated with the evaluation of the data from ESSAP's single sampling event in relation to Rockwell/Rocketdyne's ongoing monitoring data, as well as, expected natural fluctuations in activity levels, the observed H-3 activity levels are comparable.

SUMMARY

The Environmental Survey and Site Assessment Program of the Oak Ridge Institute for Science and Education performed a radiological survey of the Building 059 SNAP reactor vault which included the north test cell and pipe chase room at the Santa Susana Field Laboratory located in Ventura County, California. The survey was performed at the request of the U. S. Department of Energy's Office of Environmental Restoration, Northwestern Areas Program in order to verify the current radiological status of the test cell and pipe chase room prior to the vault containing these areas being backfilled. ESSAP personnel visited the site on December 8, 1994 and performed visual inspections, surface scans, direct surface activity measurements, exposure rate measurements, groundwater sampling, and selection of locations for core sampling. Rockwell/Rocketdyne subsequently performed the core sampling and provided the cores as well as two additional groundwater samples to ESSAP for analysis.

The results of the radiological surveys confirm that Rockwell/Rocketdyne has adequately estimated the residual radionuclide activity levels in the test cell concrete and area groundwater that were used in the RESRAD pathway analysis. In addition, the current exposure rate levels were also verified. There is, however, one exception: it is unclear from the supporting documentation as to whether or not Rockwell/Rocketdyne has accounted for the residual contamination that is present within the pipe chase room for the RESRAD pathway analysis. If unaccounted for, ESSAP recommends inclusion of this additional activity for the final facility estimated dose calculations.

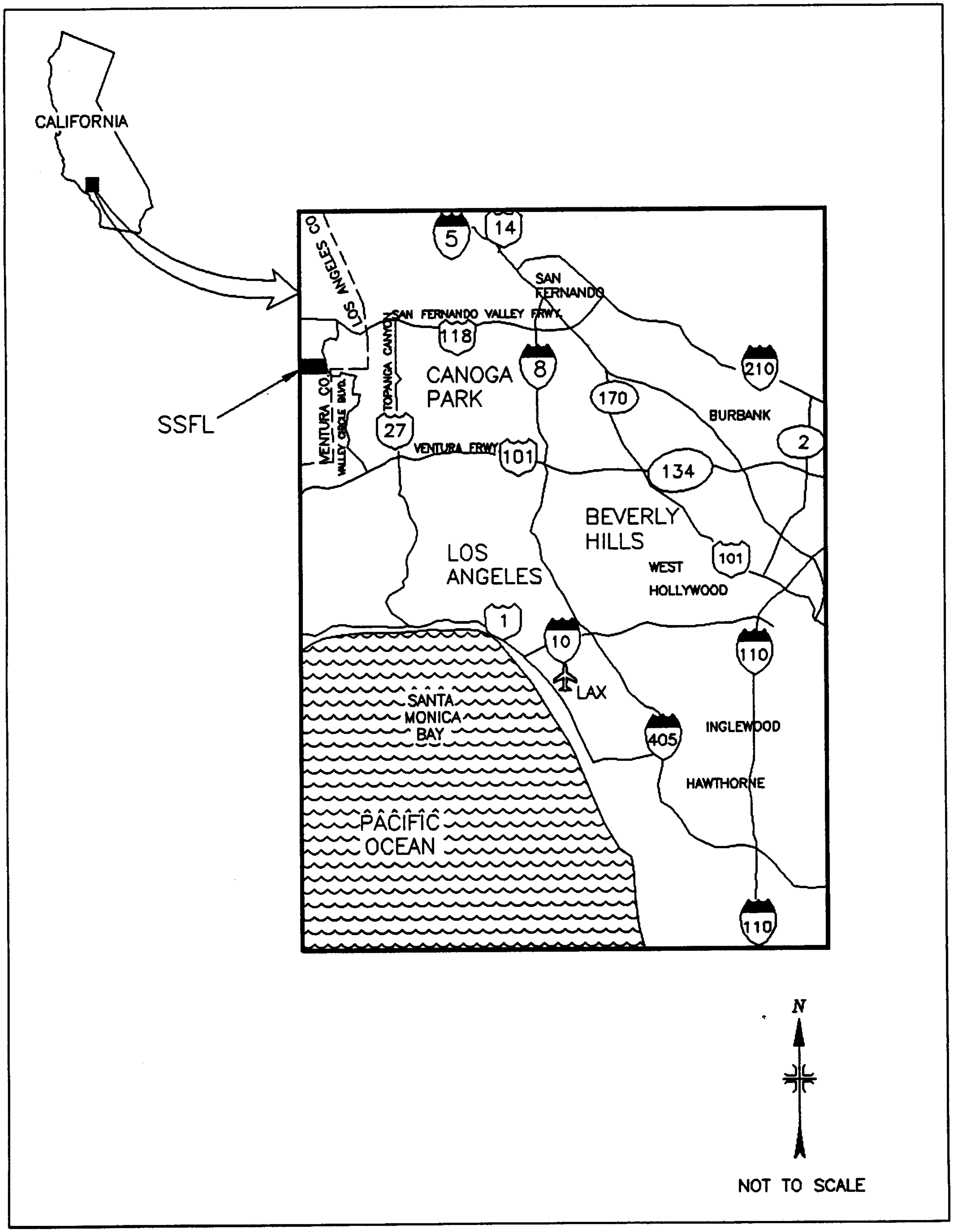

FIGURE 1: Los Angeles California Area — Location of Santa Susana Field Laboratory Site

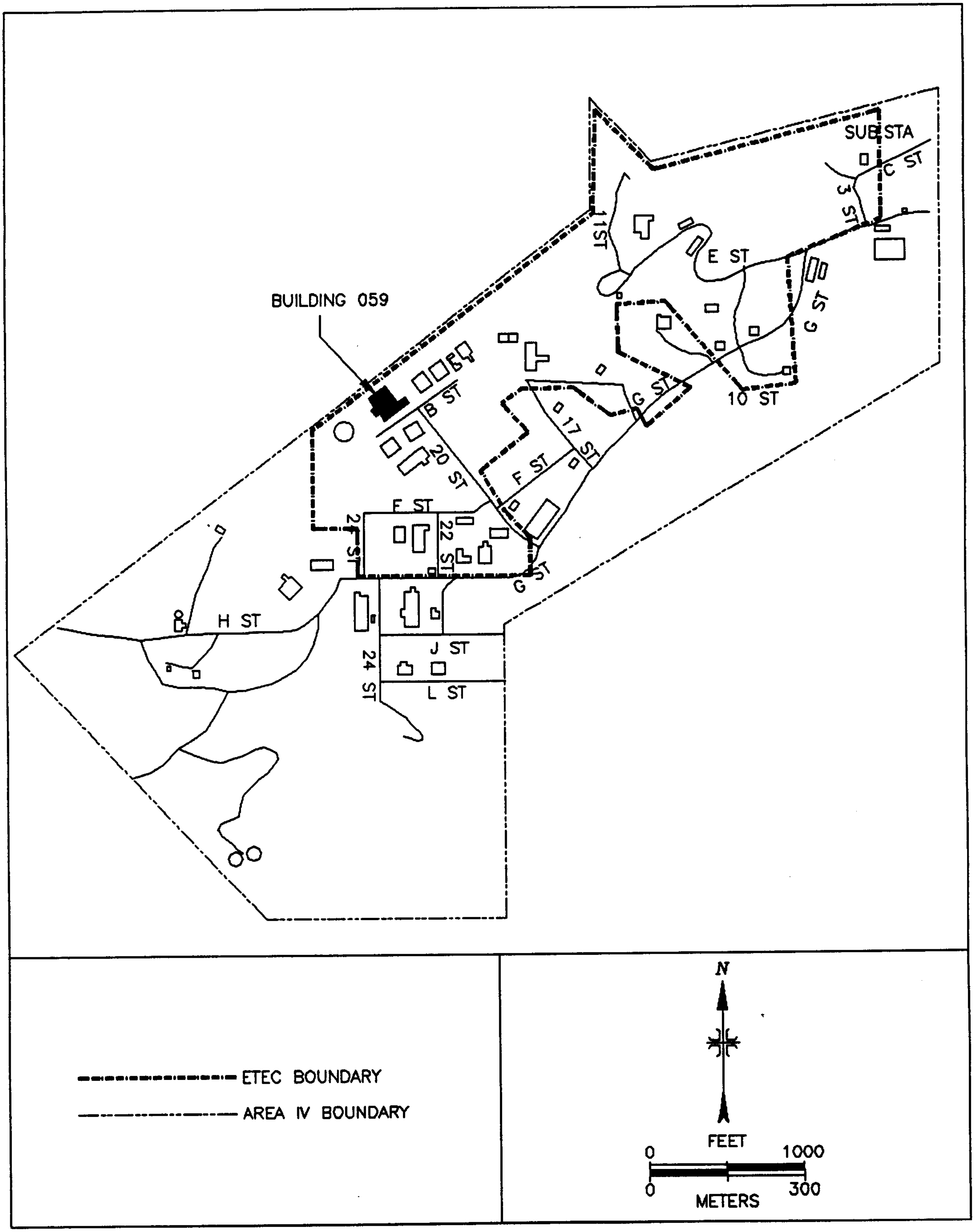

FIGURE 2: Santa Susana Field Laboratory Area IV, Plot Plan – Location of Building 059

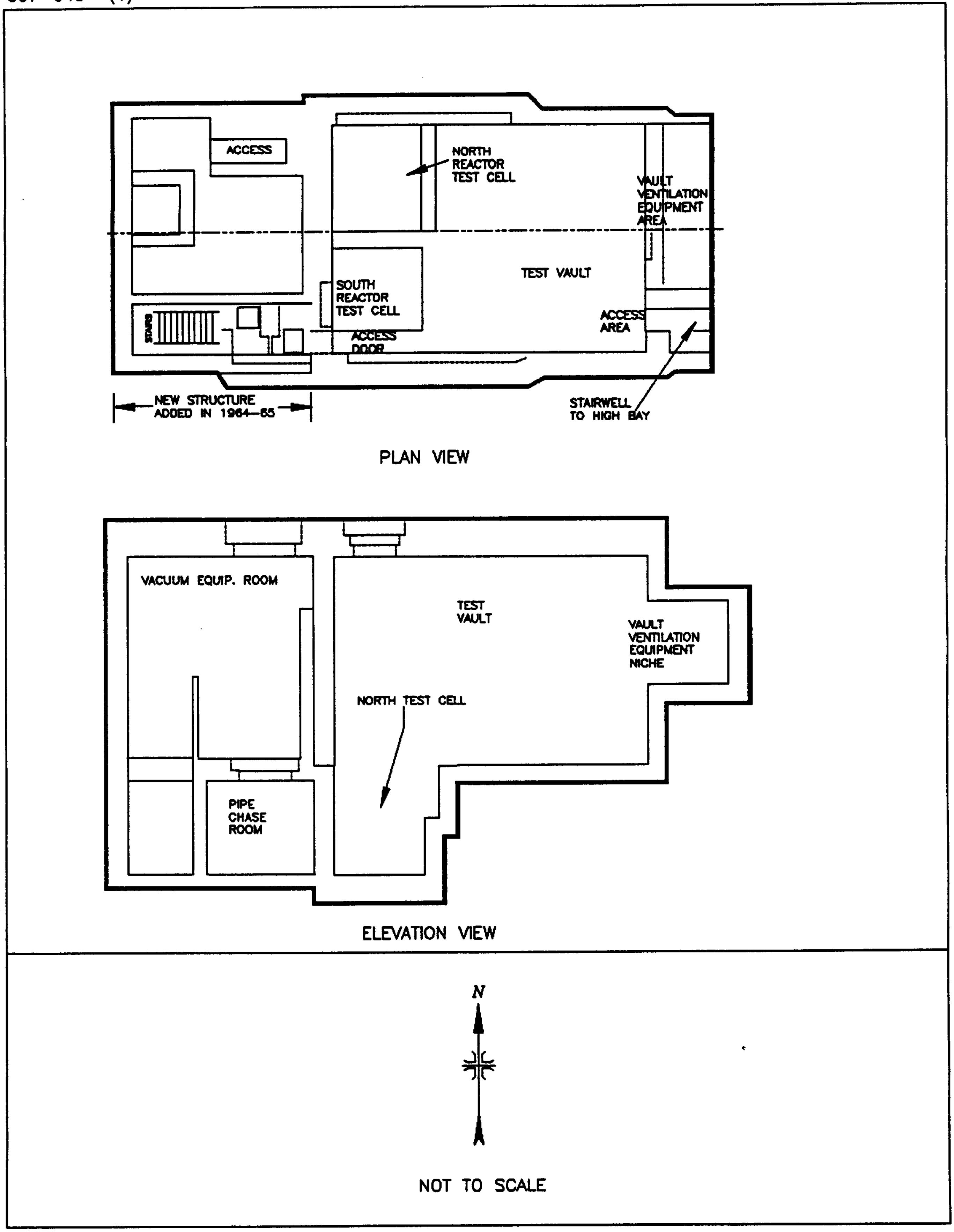

FIGURE 3: Building 059 – Location of North Test Cell and Pipe Chase Room

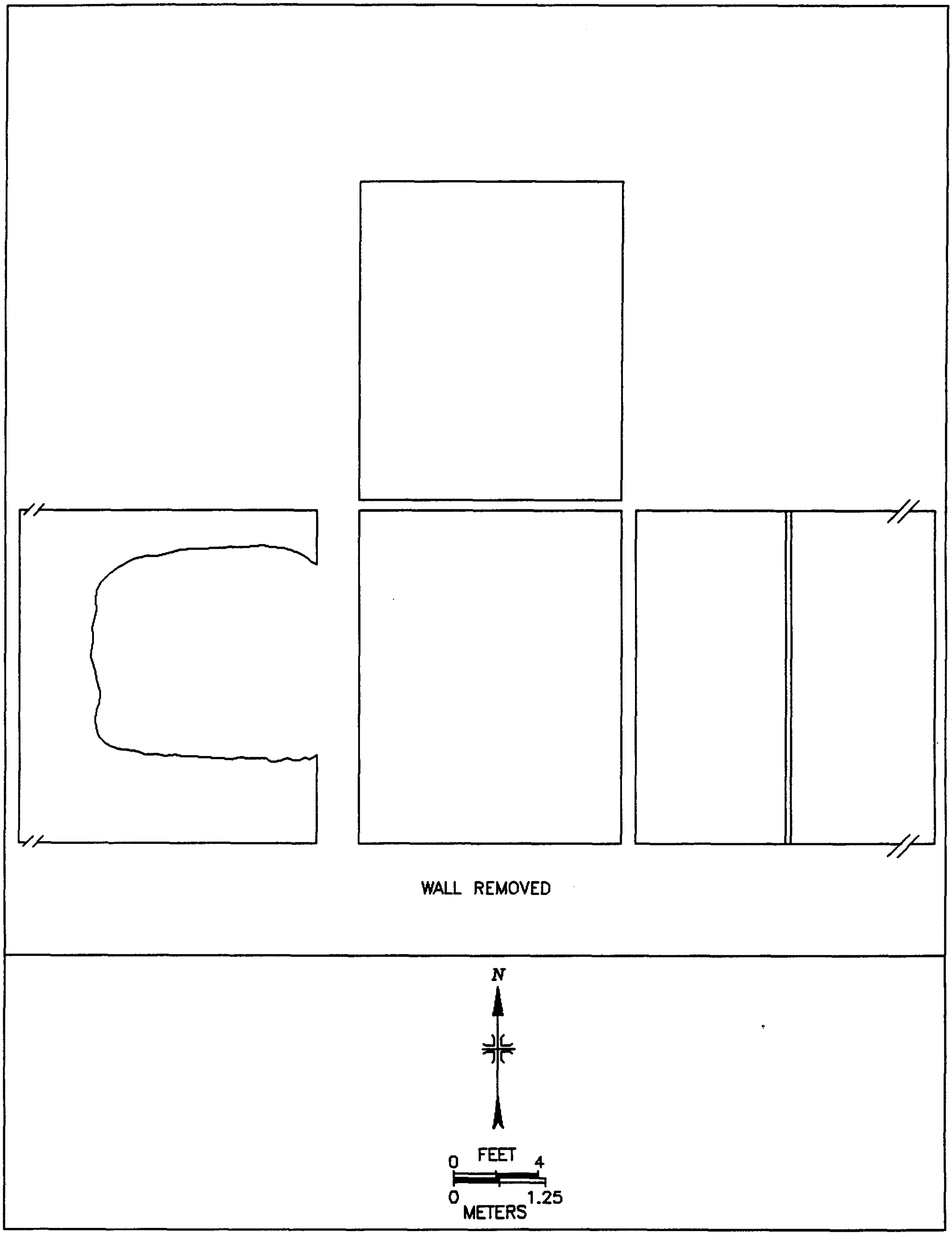

FIGURE 4: Building 059 North Test Cell – Plot Plan

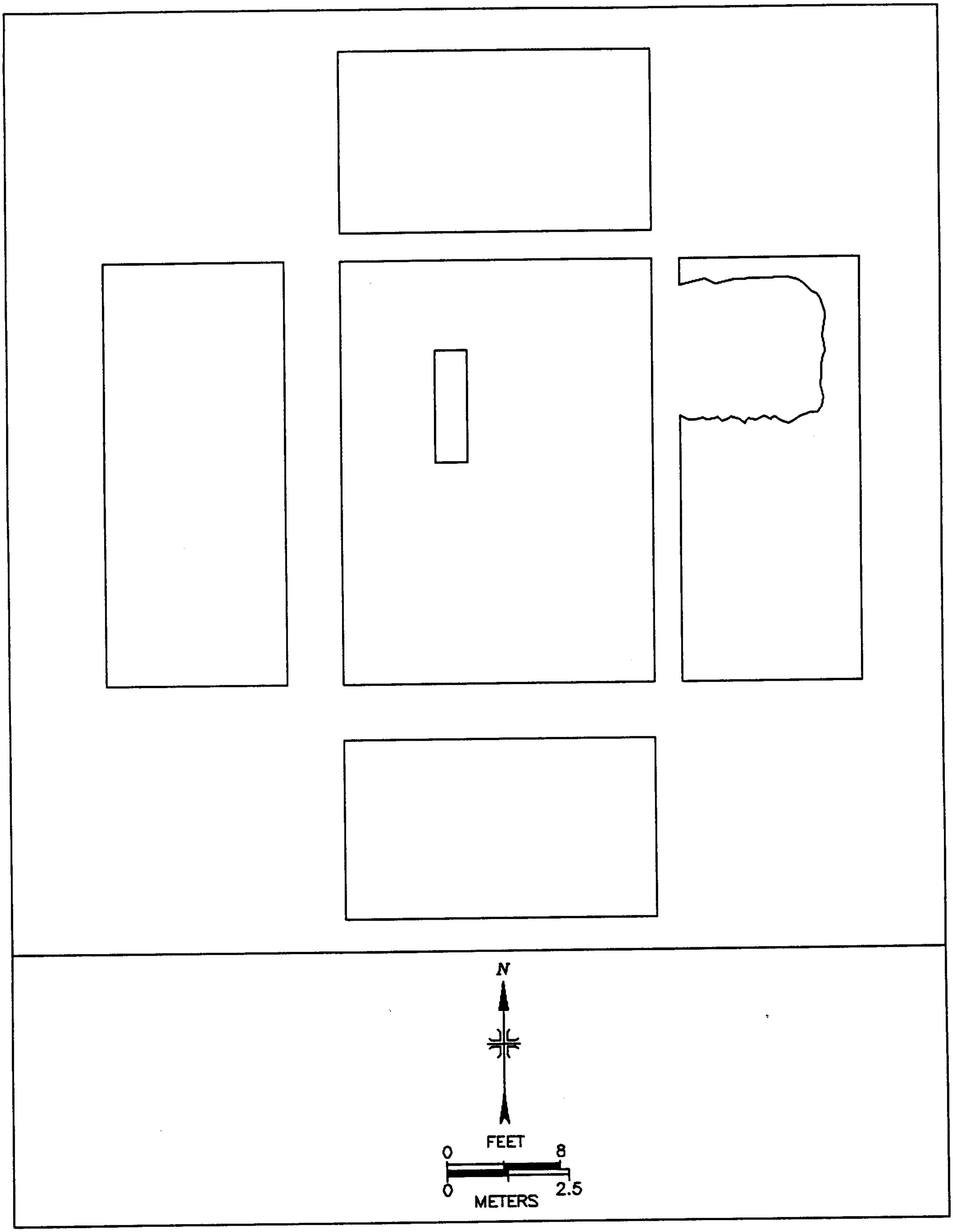

FIGURE 5: Building 059 Pipe Chase Room – Plot Plan

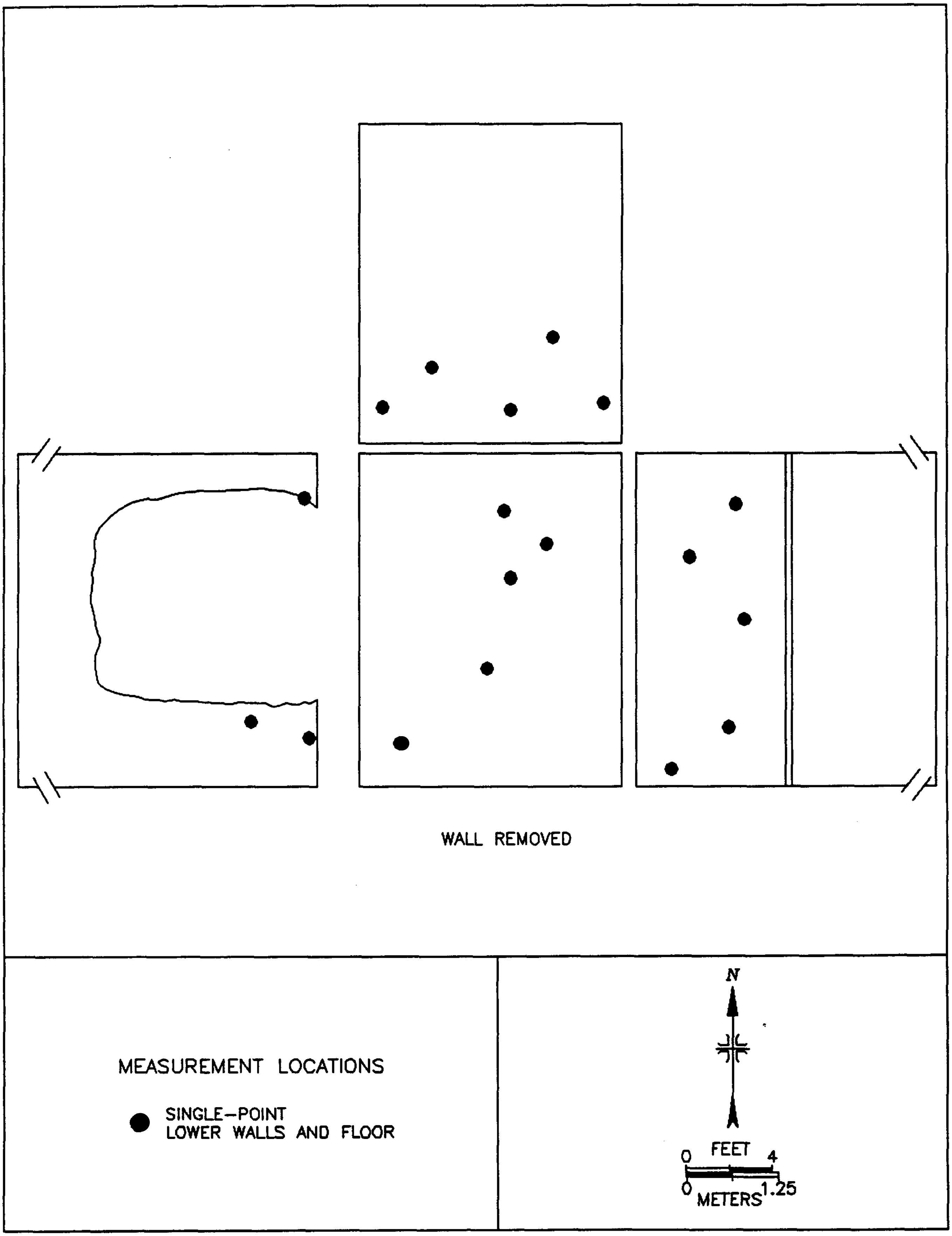

FIGURE 6: Building 059 North Test Cell — Measurement Locations

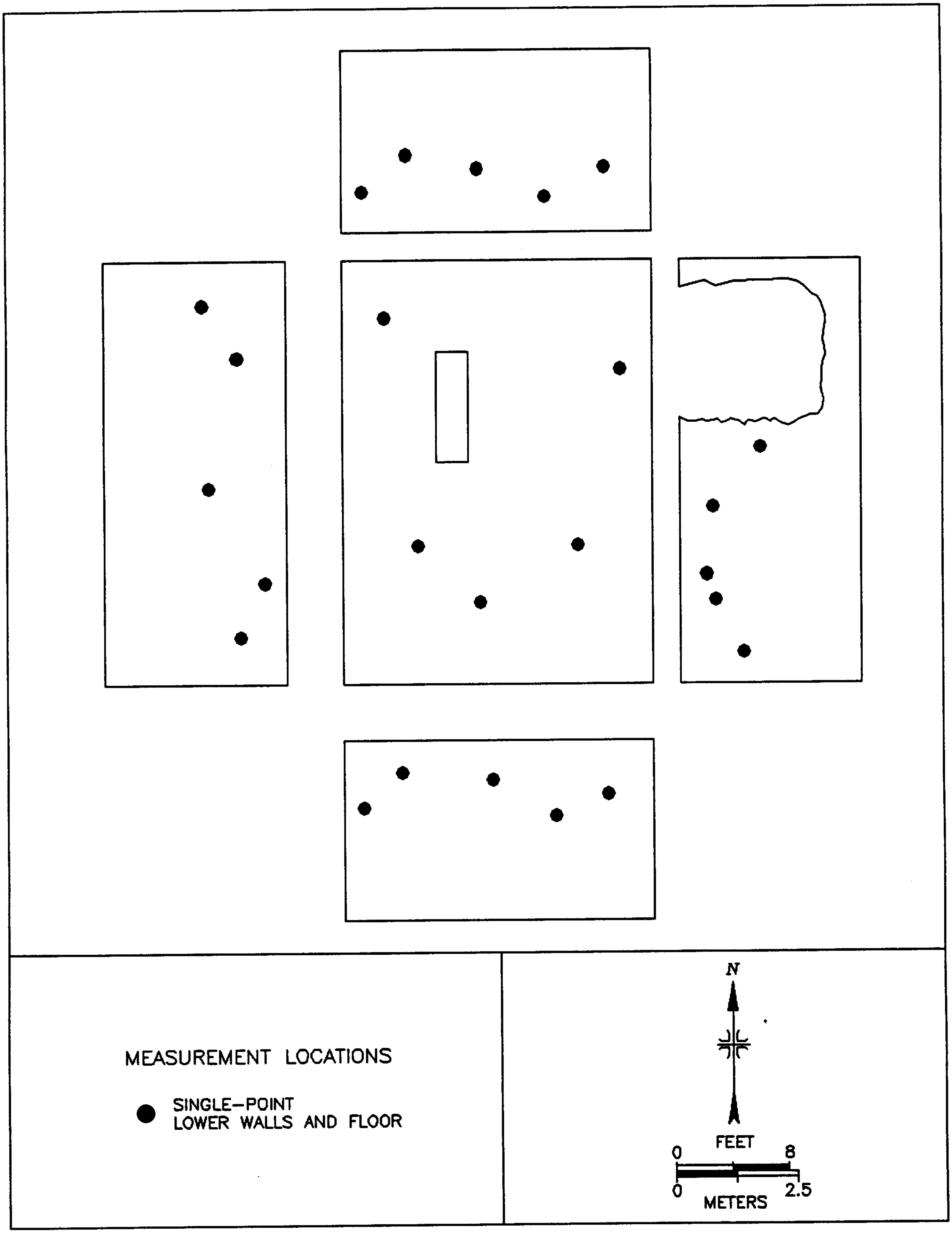

FIGURE 7: Building 059 Pipe Chase Room — Measurement Locations

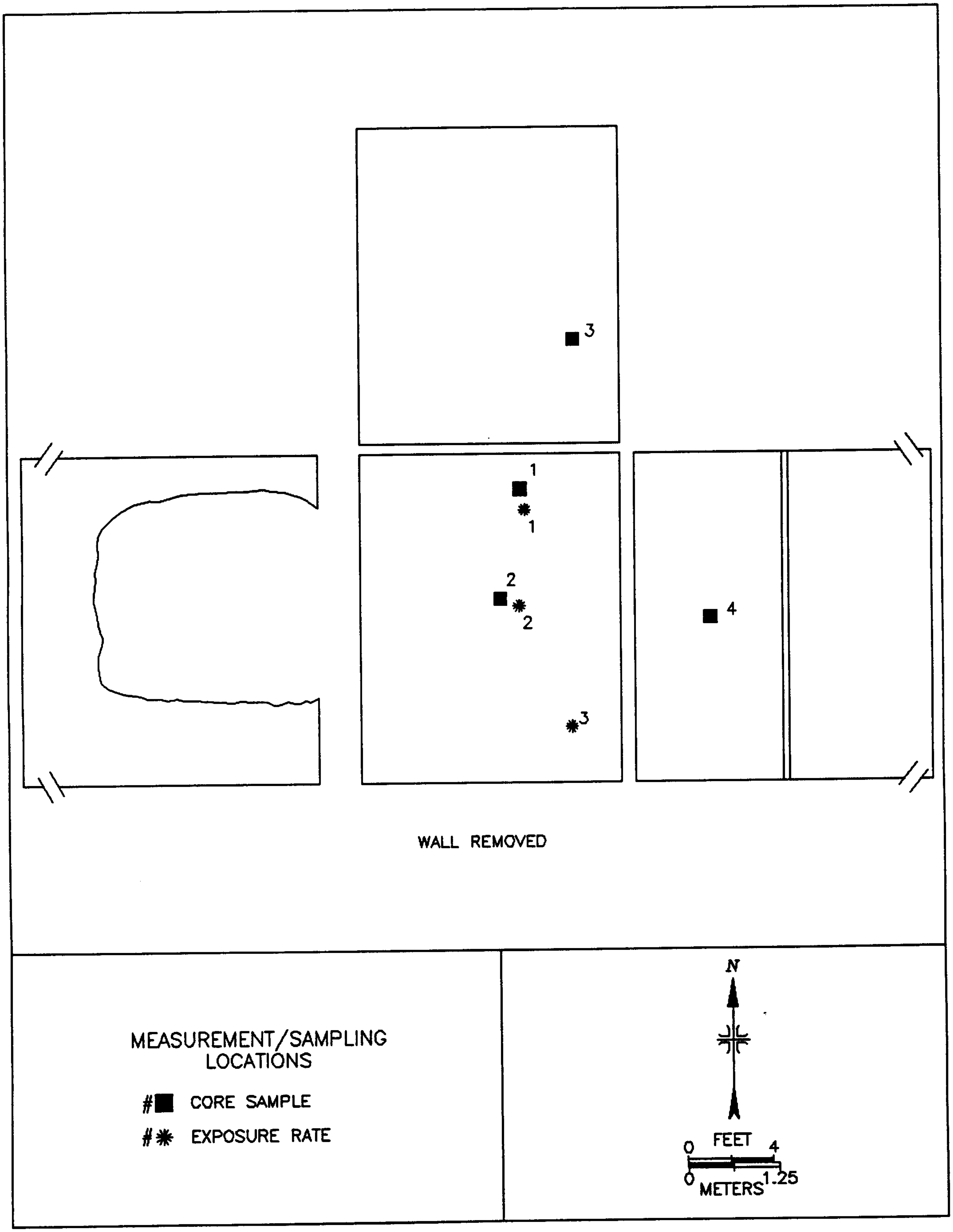

FIGURE 8: Building 059 North Test Cell — Exposure Rate Measurement and Sampling Locations

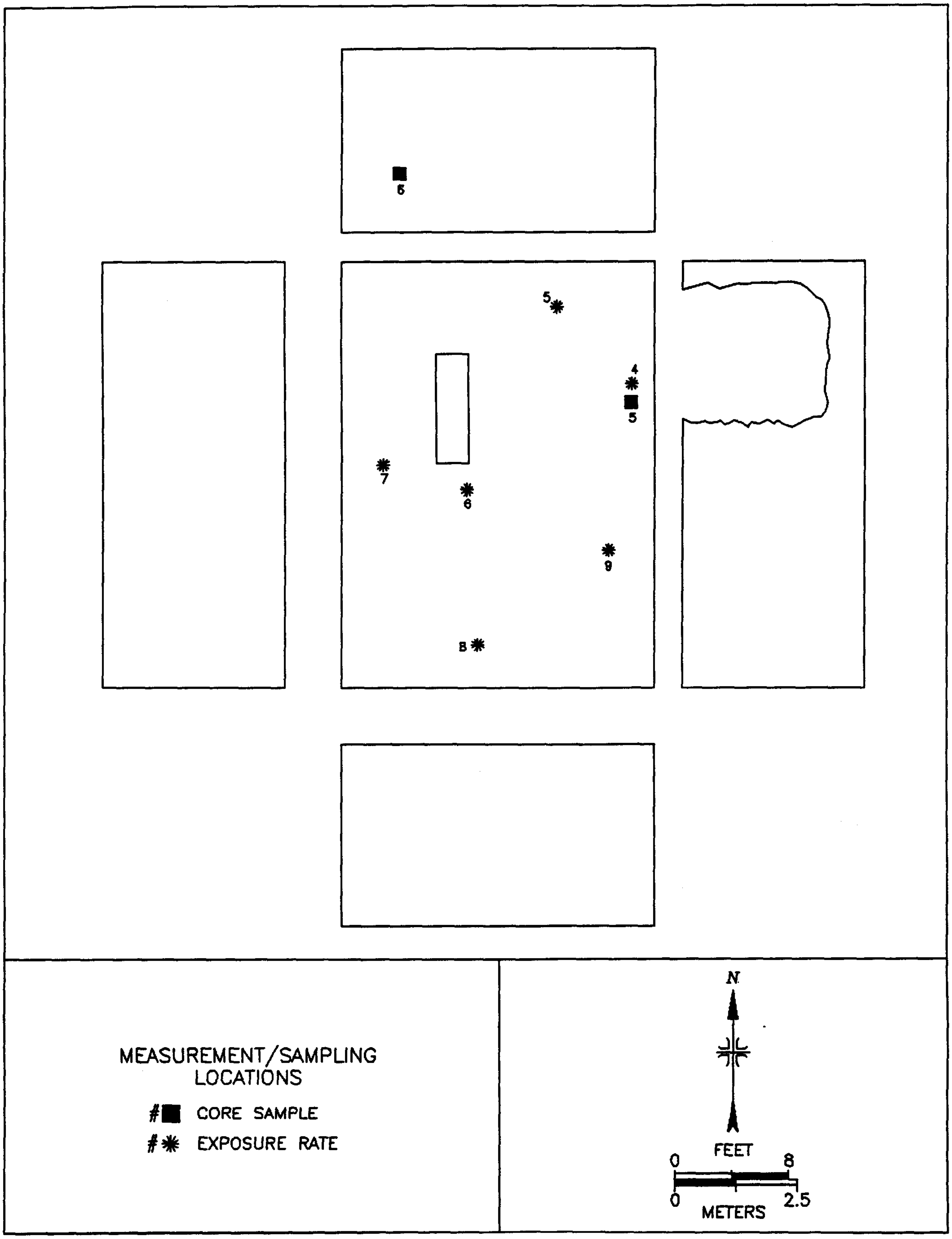

FIGURE 9: Building 059 Pipe Chase Room — Exposure Rate Measurement and Sampling Locations

TABLE 1

SUMMARY OF SURFACE ACTIVITY LEVELS AND EXPOSURE RATES
BUILDING 059 REACTOR VAULT
SANTA SUSANA FIELD LABORATORY
ROCKWELL INTERNATIONAL
VENTURA COUNTY, CALIFORNIA

Location[a]	Number of Single Point Measurements	Total Beta activity Range (dpm/100 cm^2)	Number of Exposure Rate Measurements	Exposure Rate Range (mR/h)
North Test Cell				
Floor	5	81,000 to 4,400,00	3	2.6 to 5.5
Lower Walls	13	32,000 to 190,000	NA	NA
Pipe Chase Room				
Floor	5	38,000 to 1,700,000	6	0.7 to 4.0
Lower Walls	20	9,000 to 2,600,000	NA	NA

[a]Refer to Figures 6 through 9.

TABLE 2

RADIONUCLIDE CONCENTRATIONS IN CONCRETE SAMPLES
BUILDING 059 REACTOR VAULT
SANTA SUSANA FIELD LABORATORY
ROCKWELL INTERNATIONAL
VENTURA COUNTY, CALIFORNIA

Sample ID	Location[a]		Radionuclide Concentration (pCi/g)					
	Core #	Depth (cm)	Co-60	Co-58	Ba-133	Eu-152	Eu-154	Eu-155
North Test Cell								
21784	1	2.5	3,580 ± 32[b]	157 ± 16	323 ± 15	42,700 ± 90	3,340 ± 81	35 ± 19
21785	1	5.1	1,000 ± 13	30.6 ± 6.0	107.0 ± 5.8	8,260 ± 32	1,050 ± 31	18.3 ± 6.6
21786	1	7.6	671 ± 10	17.9 ± 4.3	87.1 ± 4.9	4,380 ± 23	836 ± 24	14.9 ± 5.6
21787	1	10.2	457 ± 13	5.6 ± 4.9	55.0 ± 4.6	1,950 ± 25	469 ± 31	6.6 ± 6.9
21788	1	12.7	250.0 ± 9.0	7.2 ± 3.3	29.3 ± 3.6	1,500 ± 19	381 ± 23	10.0 ± 4.6
21789	1	15.2	230.0 ± 8.8	<5.2	18.2 ± 3.9	1,710 ± 22	412 ± 22	10.8 ± 5.8
21790	1	27.9	26.6 ± 3.2	<2.3	<2.5	149.0 ± 7.3	29 ± 12	<3.4
21791	1	63.5	<0.7	<0.6	<0.7	<2.1	<3.2	<1.2
21792	1	118.7	<0.7	<0.3	<0.4	<0.9	<1.9	<0.6
21793	2	2.5	638 ± 13	17.2 ± 5.5	63.9 ± 5.7	3,610 ± 29	758 ± 31	14.8 ± 7.1
21795	2	5.1	427 ± 13	<19.2	45.2 ± 6.4	2,050 ± 28	450 ± 34	<13.3
21796	2	7.6	287.0 ± 9.3	5.2 ± 3.3	7.1 ± 4.7	1,540 ± 19	323 ± 24	7.1 ± 4.7
21797	2	10.2	234.0 ± 9.8	<6.6	22.3 ± 4.0	1,250 ± 21	264 ± 27	<10
21798	2	12.7	130.0 ± 6.0	<3.5	13.1 ± 2.4	754 ± 13	158 ± 16	<4.4
21799	2	15.2	87.3 ± 4.4	<2.5	6.7 ± 2.4	476 ± 11	104 ± 12	<3.4
21800	2	27.9	6.1 ± 1.8	<1.4	<1.5	43.9 ± 4.1	<9.7	<2.0

TABLE 2 (Continued)

RADIONUCLIDE CONCENTRATIONS IN CONCRETE SAMPLES
BUILDING 059 REACTOR VAULT
SANTA SUSANA FIELD LABORATORY
ROCKWELL INTERNATIONAL
VENTURA COUNTY, CALIFORNIA

Sample ID	Location[a]		Radionuclide Concentration (pCi/g)					
	Core #	Depth (cm)	Co-60	Co-58	Ba-133	Eu-152	Eu-154	Eu-155
21801	2	63.5	<0.7	<0.4	<0.5	<1.4	<2.0	<0.8
21802	2	111.1	<0.4	<0.4	<0.4	<1.2	<1.4	<0.5
21803	3	2.5	101.0 ± 4.7	<1.6	<1.4	59.0 ± 4.5	18.7 ± 7.4	<1.6
21804	3	5.1	10.2 ± 1.6	<1.0	1.9 ± 1.0	55.7 ± 3.8	<6.7	<1.5
21805	3	7.6	7.2 ± 1.8	<1.5	<1.6	41.5 ± 4.0	<9.8	<2.1
21806	3	10.2	4.6 ± 1.4	<0.7	<0.8	30.2 ± 2.6	<5.4	<1.0
21807	3	12.7	5.8 ± 1.4	<0.7	<0.8	24.8 ± 2.7	7.1 ± 3.6	<1.1
21808	3	15.2	2.0 ± 1.0	<0.6	<0.6	13.6 ± 2.0	<4.2	<0.8
21809	3	27.9	<1.7	<0.9	<0.9	<3.9	<3.8	<1.3
21810	3	63.5	<0.6	<0.4	<0.4	<0.9	<1.9	<0.5
21811	3	76.2	<1.0	<0.4	<0.4	6.5 ± 1.5	<2.2	<0.7
21812	4	2.5	<1.2	<0.9	<1.1	<2.6	<2.8	<1.3
21813	4	5.1	<0.9	<0.5	<0.6	<1.5	<2.5	<0.7
21814	4	7.6	<0.8	<0.5	<0.4	<1.3	<1.9	<0.6
21815	4	10.2	<1.0	<0.8	<0.6	<1.8	<3.2	<1.2
21816	4	12.7	<0.6	<0.3	<0.4	<1.0	<1.8	<0.5
21817	4	15.2	<0.6	<0.3	<0.5	<1.0	<2.4	<0.6

TABLE 2 (Continued)

RADIONUCLIDE CONCENTRATIONS IN CONCRETE SAMPLES
BUILDING 059 REACTOR VAULT
SANTA SUSANA FIELD LABORATORY
ROCKWELL INTERNATIONAL
VENTURA COUNTY, CALIFORNIA

Sample ID	Location[a]		Radionuclide Concentration (pCi/g)					
	Core #	Depth (cm)	Co-60	Co-58	Ba-133	Eu-152	Eu-154	Eu-155
21818	4	27.9	<0.7	<0.3	<0.5	<1.0	<2.0	<0.6
21819	4	63.5	<1.2	<0.9	<0.9	<2.6	<2.0	<1.0
21820	4	73.7	<1.0	<0.3	<0.8	<1.9	<2.8	<1.0
Pipe Chase Room								
21821	5	2.5	770 ± 17	36.4 ± 8.0	<9.1	10,100 ± 47	<57	<15
21822	5	5.1	756 ± 23	49 ± 11	39.2 ± 9.4	12,700 ± 71	623 ± 59	<29
21823	5	7.6	505 ± 18	29.9 ± 9.5	<9.5	7,670 ± 50	353 ± 44	<15
21825	5	10.2	858 ± 19	37 ± 11	<8.2	6,430 ± 44	276 ± 44	<13
21826	5	12.7	662 ± 15	21.0 ± 6.3	<7.3	5,610 ± 38	<45	<12
21827	5	15.2	245 ± 13	16.9 ± 7.5	<10.2	4,340 ± 43	173 ± 33	<18
21828	5	27.9	49.3 ± 6.3	<4.1	<3.4	952 ± 17	<20	<5.0
21829	5	63.5	<1.1	<0.7	<0.6	<2.8	<3.0	<0.8
21830	5	86.4	<1.3	<0.8	<0.9	<2.5	<4.7	<1.3
21831	6	2.5	1,340 ± 18	<5.3	<3.4	<7.3	<21	<3.3
21832	6	5.1	1.5 ± 1.0	<0.6	<0.8	<2.3	<3.4	<1.0
21833	6	7.6	<1.2	<0.5	<0.5	<1.1	<1.3	<0.6
21834	6	10.2	<1.2	<0.4	<0.5	<1.3	<2.3	<0.8

TABLE 2 (Continued)

RADIONUCLIDE CONCENTRATIONS IN CONCRETE SAMPLES
BUILDING 059 REACTOR VAULT
SANTA SUSANA FIELD LABORATORY
ROCKWELL INTERNATIONAL
VENTURA COUNTY, CALIFORNIA

Sample ID	Location[a]		Radionuclide Concentration (pCi/g)					
	Core #	Depth (cm)	Co-60	Co-58	Ba-133	Eu-152	Eu-154	Eu-155
21835	6	12.7	<2.0	<0.7	<0.8	<2.1	<2.9	<1.0
21836	6	15.2	<1.2	<0.4	<0.4	<0.9	<1.6	<0.6
21837	6	27.9	<1.3	<0.5	<0.4	<1.1	<2.6	<0.7
21838	6	61.0	2.3 ± 1.0	<0.8	<0.7	<1.6	<2.1	<1.0

[a]Refer to Figures 8 and 9.
[b]Uncertainties represent the 95% confidence level, based only on counting statistics.

TABLE 3

RADIONUCLIDE CONCENTRATIONS IN WATER SAMPLES
BUILDING 059 REACTOR VAULT
SANTA SUSANA FIELD LABORATORY
ROCKWELL INTERNATIONAL
VENTURA COUNTY, CALIFORNIA

Location	Radionuclide Concentration (pCi/g)		
	Gross Alpha	Gross Beta	Tritium
West Side 475 French Drain	28.3 ± 5.7[a]	21.5 ± 4.8	<830
East Side Well 476 RD24	11.6 ± 2.3	11.6 ± 1.5	<830
Well 24 3A	3.6 ± 1.5	8.4 ± 1.4	<750
Well 28 4A	11.1 ± 2.5	14.7 ± 1.9	920 ± 470

[a]Uncertainties represent the 95% confidence level, based only on counting statistics.

REFERENCES

1. Letter from G. G. Gaylord, Rockwell International to J. Cullen, U.S. Department of Energy, "Decontamination and Decommissioning of Building 059," March 17, 1994.

2. U.S. Department of Energy, DOE/CH/8901, "A Manual for Implementing Residual Radioactive Material Guidelines," June 1989.

3. Letter from M. R. Landis, ORISE to D. Williams, U.S. Department of Energy, "Decontamination and Decommissioning (D&D) of Building 059, March 17, 1994 [94ETEC-DRF 0457]," April 1, 1994.

4. Letter from G. G. Gaylord, Rockwell International to M. Lopez, U.S. Department of Energy, "Response to Comments by ORISE on Decontamination and Decommissioning (D&D) of Building 059," June 1, 1994.

5. Letter from T. J. Vitkus, ORISE to A. Kluk, U.S. Department of Energy, "Revised Radiological Survey Plan for the Building 059 Reactor Vault, Santa Susana Field Laboratory, Ventura County, California," December 6, 1994.

APPENDIX A

MAJOR INSTRUMENTATION

APPENDIX A

MAJOR INSTRUMENTATION

The display of a specific product is not to be construed as an endorsement of the product or its manufacturer by the authors or their employers.

DIRECT RADIATION MEASUREMENT

Instruments

Bicron Micro-Rem Meter
(Bicron Corporation, Newburg, OH)

Eberline Pulse Ratemeter
Model PRM-6
(Eberline, Santa Fe, NM)

Ludlum Ratemeter-Scaler
Model 2221
(Ludlum Measurements, Inc.,
Sweetwater, TX)

Detectors

Eberline Compensated GM Detector
Model HP-270
(Eberline, Santa Fe, NM)

Eberline GM Detector
Model HP-210
Effective Area, 15.5 cm^2
(Eberline, Santa Fe, NM)

LABORATORY ANALYTICAL INSTRUMENTATION

High-Purity Germanium Detector
Model GMX-23195-S, 23% Eff.
(EG&G ORTEC, Oak Ridge, TN)
Used in conjunction with:
Lead Shield Model G-16
(Gamma Products, Palos Hills, IL) and
Multichannel Analyzer
3100 Vax Workstation
(Canberra, Meriden, CT)

Low Background Gas Proportional Counter
Model LB-5100-W
(Oxford, Oak Ridge, TN)

Tri-Carb Liquid Scintillation Analyzer
Model 1900CA
(Packard Instrument Co., Meriden, CT)

APPENDIX B

SURVEY AND ANALYTICAL PROCEDURES

APPENDIX B

SURVEY AND ANALYTICAL PROCEDURES

SURVEY PROCEDURES

Surface Scans

Surface scans were performed by passing the probes slowly over the surface; the distance between the probe and the surface was maintained at a minimum - nominally about 1 cm. Identification of the various gamma activity levels was based on increases in the audible signal from the recording and/or indicating instrument. Combinations of detectors and instruments used for the gamma scans were:

Gamma — compensated GM detector with ratemeter-scaler

The above gamma scans were further augmented by performing general area gamma surveys of the floors and walls using a microrem meter. The maximum general dose rate levels observed during these surveys were then used to assist in selecting locations for direct measurements, exposure rate measurements, and sampling.

Surface Activity Measurements

Measurements of total beta-gamma activity levels were performed at the locations of highest elevated direct radiation on floors and walls using a GM detector coupled with a portable ratemeter-scaler. Count rates (cpm), which were integrated over 1 minute in a static position, were converted to activity levels (dpm/100 cm^2) by dividing the net rate by the 4 π efficiency and correcting for the active area of the detector. The beta-gamma activity background count rate for the GM detector was 31 cpm. The beta efficiency factor was 0.14 for the GM detector. The effective window for the GM detector was 15.5 cm^2.

Exposure Rate Measurements

Measurements of gamma exposure rates were performed using a pressurized ionization chamber and were reported in milliroentgens per hour (mR/h).

Miscellaneous Samples

Concrete Core Sampling

ESSAP selected six locations for core sampling. Rockwell personnel used a mechanical coring device to extract intact concrete cores of the test cell and pipe chase room walls and floors. Rockwell sectioned each core into 2.54 centimeter increments and shipped the concrete samples under chain-of-custody to ESSAP for analysis.

Water Sampling

Approximately one liter of water was collected from two sample locations. The samples were transferred to an appropriate container, sealed, and labeled in accordance with ESSAP survey procedures. Rockwell personnel collected two additional samples and shipped them under chain-of-custody to ESSAP for analysis.

ANALYTICAL PROCEDURES

Miscellaneous Samples

Gamma Spectrometry

Solid Samples

Samples of concrete were crushed and a portion sealed in a 20 milliliter petri dish. The quantity placed in the container was chosen to reproduce the calibrated counting geometry and ranged from 14 to 23 g of material. Net material weights were determined and the samples counted for one hour. Background and Compton stripping, peak search, peak identification, and concentration calculations were performed using the computer capabilities inherent in the analyzer system, supplied by Canberra Nuclear Products. Energy peaks used for determination of radionuclides of concern were:

Ba-133	0.356 MeV
Co-58	0.810 MeV
Co-60	1.173 MeV
Eu-152	0.344 MeV
Eu-154	0.723 MeV
Eu-155	0.105 MeV

Spectra were also reviewed for other identifiable photopeaks.

The gamma spectrometry systems are n-type high purity germanium or n-type extended range detectors with thin beryllium windows, coupled to a pulse height analyzer system. Active detector volumes ranged from 111.9 to 133 cubic centimeters. Measured relative efficiencies ranged from 25.5 to 30.2%.

Liquid Samples

Liquid samples with a high solid content were dried, mixed, and a portion sealed in 0.5 liter (0.53 qt) Marinelli beaker. Those with a low solid content were placed in 0.5 liter containers without processing. The samples were then treated as a solid sample.

Gross Alpha and Beta in Water

A known volume of water was acidified with dilute nitric acid, concentrated and dried in a planchet. Samples were counted in a low-background proportional counter.

Tritium

Water samples were distilled to remove other potential radionuclides and organic materials. The samples were spiked with a standard tritium solution to evaluate quenching and counted in a liquid scintillation counter.

UNCERTAINTIES AND DETECTION LIMITS

The uncertainties associated with the analytical data presented in the tables of this report represent the 95% confidence level for that data. These uncertainties were calculated based on both the gross sample count levels and the associated background count levels. When the net sample count was less than the 95% statistical deviation of the background count, the sample concentration was reported as less than the detection limit of the measurement procedures. Because of variations in background levels, measurement efficiencies, and contributions from other radionuclides in samples, the detection limits differ from sample to sample and instrument to instrument. Additional uncertainties, associated with sampling and measurement procedures, have not been propagated into the data presented in this report.

CALIBRATION AND QUALITY ASSURANCE

Calibration of all field and laboratory instrumentation was based on standards/sources, traceable to NIST, when such standards/sources were available. In cases where they were not available, standards of an industry recognized organization were used. Calibration of pressurized ionization chambers was performed by the manufacturer.

Analytical and field survey activities were conducted in accordance with procedures from the following documents:

- Survey Procedures Manual, Revision 8 (December 1993)
- Laboratory Procedures Manual, Revision 8 (August 1993) and Revision 9 (January 1995)
- Quality Assurance Manual, Revision 6 (July 1993)

The procedures contained in these manuals were developed to meet the requirements of DOE Order 5700.6C and ASME NQA-1 for Quality Assurance and contain measures to assess processes during their performance.

Quality control procedures include:

- Daily instrument background and check-source measurements to confirm that equipment operation is within acceptable statistical fluctuations.
- Participation in EPA and EML laboratory Quality Assurance Programs.
- Training and certification of all individuals performing procedures.
- Periodic internal and external audits.